Tennessee Valley Authority, Post Office 2000, Spring City, Tennessee 37381-2000

Mike Skaggs
Site Vice President, Watts Bar Nuclear Plant

SEP 26 2006

10 CFR 50.73

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of) Docket No. 50-390
Tennessee Valley Authority)

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - LICENSEE EVENT REPORT 390/2006-005 - REACTOR TRIP DUE TO MAIN GENERATOR TRIP

This submittal provides LER 390/2006-005. This LER documents a event that occurred on July 31, 2006, involving an automatic reactor trip due to a main generator trip. The report contains information regarding this event is provided in accordance with 10 CFR 50.73(a)(2)(iv)(A).

There are no regulatory commitments associated with this letter. Should there be questions regarding this submittal, please contact James D. Smith at (423) 365-1824.

Sincerely,

M Skaggs

Mike Skaggs

Enclosure
cc: See Page 2

IE22
Designated original per Doug Pickett PM

SEP 2 6 2006

Enclosure
cc (Enclosure):
NRC Resident Inspector
Watts Bar Nuclear Plant
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NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007

Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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4. TITLE

Automatic Reactor Trip Due to Generator Trip

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
07	31	2006	2006	- 005 -	000	09	26	2006	FACILITY NAME	DOCKET NUMBER
										05000

9. OPERATING MODE

1

10. POWER LEVEL

100%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

- ☐ 20.2201(b)
- ☐ 20.2201(d)
- ☐ 20.2203(a)(1)
- ☐ 20.2203(a)(2)(i)
- ☐ 20.2203(a)(2)(ii)
- ☐ 20.2203(a)(2)(iii)
- ☐ 20.2203(a)(2)(iv)
- ☐ 20.2203(a)(2)(v)
- ☐ 20.2203(a)(2)(vi)
- ☐ 20.2203(a)(3)(i)
- ☐ 20.2203(a)(3)(ii)
- ☐ 20.2203(a)(4)
- ☐ 50.36(c)(1)(i)(A)
- ☐ 50.36(c)(1)(ii)(A)
- ☐ 50.36(c)(2)
- ☐ 50.46(a)(3)(ii)
- ☐ 50.73(a)(2)(i)(A)
- ☐ 50.73(a)(2)(i)(B)
- ☐ 50.73(a)(2)(i)(C)
- ☐ 50.73(a)(2)(ii)(A)
- ☐ 50.73(a)(2)(ii)(B)
- ☐ 50.73(a)(2)(iii)
- ☑ 50.73(a)(2)(iv)(A)
- ☐ 50.73(a)(2)(v)(A)
- ☐ 50.73(a)(2)(v)(B)
- ☐ 50.73(a)(2)(v)(C)
- ☐ 50.73(a)(2)(v)(D)
- ☐ 50.73(a)(2)(vii)
- ☐ 50.73(a)(2)(viii)(A)
- ☐ 50.73(a)(2)(viii)(B)
- ☐ 50.73(a)(2)(ix)(A)
- ☐ 50.73(a)(2)(x)
- ☐ 73.71(a)(4)
- ☐ 73.71(a)(5)
- ☐ OTHER

Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
Rickey Stockton, Licensing Engineer	(423) 365-1818

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED

○ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ◉ NO

15. EXPECTED SUBMISSION DATE

MONTH	DAY	YEAR

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On July 31, 2006, at approximately 12:13 EDT, the control room unexpectedly received an exciter field overcurrent alarm, followed immediately thereafter (< 1 sec) by the generator field breaker opening, and a turbine/reactor trip.

An event team was assembled subsequent to the plant trip. The most likely cause was determined to be in the main generator automatic excitation control circuitry. Corrective action taken was to return the plant to service while monitoring several input and output signals associated with the automatic controls so that any subsequent events could be captured and analyzed. The excitation system would also be operated in TEST mode instead of AUTOMATIC to preclude the possibility of an additional plant trip.

Since restarting the plant, there have been several instances where the output of the Maximum Excitation Limiter (MXL) circuit board in the automatic excitation control circuitry has changed significantly (+14 volts dc to -15 volts dc) with no corresponding change on any of the inputs. Had the automatic circuit been in service, the MXL output change would have driven the excitation controls into a loss of the exciter field which was the case during the plant trip. Based on this data, during the current refueling outage, the MXL circuit board will be removed to determine if a discreet component on the device has failed and an inspection performed of the interface wiring to the MXL circuit board to determine if degraded wiring caused the failure.

As a result of the plant trip, the actuation of the Reactor Protection and the Auxiliary Feedwater Systems were reported in accordance with 10 CFR 50.72(b)(2)(iv) and 10 CFR 50.72(b)(3)(iv), respectively. This event is also being reported as this Licensee Event Report in accordance with 10 CFR 50.73 (a)(2)(iv).

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

I. Plant Conditions:

On July 31, 2006, WBN Unit 1 was in Mode 1 at approximately 100 percent reactor thermal power. The operating temperature was 588 degrees F and Reactor Coolant System (RCS) (Energy Industry Identification System (EIIS) Code AB) pressure was 2235 psig.

II. Description of Event:

A. Event:

On July 31, 2006 at approximately 12:13 EDT, the control room unexpectedly received an exciter field overcurrent alarm, followed immediately thereafter (< 1 sec) by the generator field breaker opening, and a turbine/reactor trip. Control rods (EIIS Code ROD) inserted and the Auxiliary Feedwater System (EIIS code BA) auto started as designed. The plant was stabilized in Mode 3. The plant trip was normal except for Reactor Coolant Pump (RCP) No. 2 (EIIS Code AB/P) which did not auto transfer back to its start bus (EIIS Code BU).

As a result of the plant trip, the actuation of the Reactor Protection (EIIS code JC) and the Auxiliary Feedwater Systems were reported in accordance with 10 CFR 50.72(b)(2)(iv) and 10 CFR 50.72(b)(3)(iv), respectively. This event is also being reported as this Licensee Event Report in accordance with 10 CFR 50.73 (a)(2)(iv).

B. Inoperable Structures, Components, or Systems that Contributed to the Event

There were no additional structures, components or systems inoperable at the start of the event that contributed to the event.

C. Dates and Approximate Times of Major Occurrences

Date	Time	Event
July 31, 2006	- 12:12	Unit 1 Reactor at Full Power - All conditions normal
	- 12:13	Exciter Field Overcurrent Alarm
	- 12:13	Reactor Trip/Turbine Trip
	- 12:13	Generator Trip
	- 12:13	Entered in Mode 3

D. Other Systems or Secondary Functions Affected

The plant trip was normal except for Reactor Coolant Pump (RCP) No. 2 which did not auto transfer back to its start bus. A manual attempt at reclosure by the operator was unsuccessful. Investigation found, based on interview of the craft performing the repair work, that the "anti pumping" relay (52Y) on the RCP supply breaker (EIIS Code BKR) did not have the minimum spacing between the armature tail piece and the molded posts to assure free movement of the armature. The craft corrected this problem by providing more clearance between the contact mounting posts and the armature. Contact wear may also be a contributing factor. Subsequently, RCP No. 2 was returned to service.

E. Method of Discovery

As described above, this condition was first identifed when the Exciter Field Overcurrent Alarm sounded.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

F. Operator Actions

Crew response to the event was timely and met Operations and Training management expectations. There were no human performance issues.

G. Safety System Responses

Upon reactor trip, Auxiliary Feedwater System started as designed.

III. CAUSE OF EVENT

The Kepner-Tregoe (KT) analysis indicated that the most likely cause of the trip was an intermittent failure in the automatic controls of the excitation system. The most likely cause was determined to be a failed Maximum Excitation Limiter (MXL) circuit board in the main generator automatic excitation control circuitry. Data obtained from the plant annunciator system and the main generator protective devices from the time of the trip indicate that the plant tripped due to a loss of the main generator exciter field.

IV. ANALYSIS OF THE EVENT

Plant safety systems functioned normally in response to the reactor trip. All rods inserted fully. Steam generator level was maintained initially via the Auxiliary Feedwater system. See Section V, "Assessment of Safety Consequences," below for further discussion.

V. ASSESSMENT OF SAFETY CONSEQUENCES

The automatic reactor trip on July 31, 2006 can be compared to the FSAR Loss of External Electrical Load and/or Turbine Trip in Update Final Safety Analysis Report section 15.2.7 (page 15.2-21). The automatic trip at approximately 12:13 occurred as a result of secondary side electrical problems. The alarm printout indicated exciter overcurrent which may have been due to either problems in the voltage regulator circuit or a fault in the generator bus to the main transformers. The plant trip was normal except for RCP No. 2 did not auto transfer to the start bus. A manual attempt at reclosure was unsuccessful and the RCP No. 2 breaker was subsequently repaired. The plant was stabilized using steam dumps. The reactor coolant system responded to the initial transient as expected with no pressurizer power operated relief valve (PORV) (EIIS Code PCV) actuation, no safety injection initiation, and no steam generator atomspheric dump valve actuation.

The UFSAR 15.2.7 analysis contains several analysis conservatisms which were not characteristic of the actual event. The UFSAR analysis assumes the reactor trip is based on a reactor protection system trip setpoint exceedance and in this case was initiated by a secondary side turbine trip. In addition, reactor control is assumed to be in manual, no credit is taken for the steam dump system, and no credit is taken for the steam generator atmospheric relief valves (only steam generator safeties are credited). The actual event had automatic rod control and steam dumps available. The UFSAR analysis demonstrates for two cases (DNB case where credit is taken for the pressurizer PORVs and spray, and RCS overpressure case where no credit is taken for the pressurizer spray or PORVs) that the minimum DNBR is well above the limiting value and that the RCS pressure safety analysis limits are met.

Therefore based upon the above, the actual event is bounded by the UFSAR safety analysis assumptions.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

V. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

Operators responded to the plant transient in accordance with appropriate plant procedures. An event team was assembled to investigate the cause of the event.

B. Corrective Actions to Prevent Recurrence (TVA does not consider these items to constitute regulatory commitments. TVA's corrective action program tracks completion of these actions.)

Initial actions taken to identify the cause included: (1) a complete checkout of the voltage regulator per vendor instructions, (2) a crawl through inspection of the isophase bus, (3) megger testing of the main generator and exciter, (4) inspection and testing of the main generator potential transformer and associated fuses, (5) testing of the exciter ground detector circuit, (6) electromagnetic and radio frequency interference diagnostic testing to detect intermittent arcing, and (7) various informal walk downs. None of these actions identified a failed component.

With the most likely cause determined to be in the main generator automatic excitation control circuitry, a decision was made to return the plant to service and to monitor several input and output signals associated with the automatic controls so that any subsequent events could be captured and analyzed. The excitation system would also be operated in TEST mode instead of AUTOMATIC to preclude the possibility of an additional plant trip. Since restarting the plant, there have been several instances where the output of the Maximum Excitation Limiter (MXL) circuit board in the automatic excitation control circuitry has changed significantly (+14 volts dc to -15 volts dc) with no corresponding change on any of the inputs. Had the automatic circuit been in service, the MXL output change would have driven the excitation controls into full buck which would ultimately result in a loss of the exciter field which was the case during the plant trip. During the current refueling outage, the MXL circuit board will be removed to determine if a discreet component on the device has failed and an inspection performed of the interface wiring to the MXL circuit board to determine if degraded wring caused the failure.

VI. ADDITIONAL INFORMATION

A. Failed Components

As discussed previously, the most likely cause was determined to be a failed Maximum Excitation Limiter (MXL) circuit board originally manufactured by Siemens-Westinghouse part of voltage regulator model WTA.

B. Previous LERs on Similar Events

A review was performed of the previous WBN Licensee Event Reports (LERs) for any events associated with generator trips involving failed electronic cards. Although WBN has experienced events associated with failed circuit cards, there were no previous LERs identified which were associated with main generator automatic excitation control circuitry which caused a turbine/reactor trip.

C. Additional Information:

None.

D. Safety System Functional Failure

This event did not involve a safety system functional failure as defined in NEI 99 02, Revision 4.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

E. Loss of Normal Heat Removal Consideration

This event is not considered a scram with loss of normal heat removal.

VII. COMMITMENTS

None